DEVELOPMENT OF IRT-TYPE FUEL ASSEMBLY WITH PIN-TYPE FUEL ELEMENTS FOR LEU CONVERSION OF WWR-SM RESEARCH REACTOR IN UZBEKISTAN

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ABSTRACT

The principle design of an IRT-type FA with pin-type fuel elements has been developed.
This work is being conducted in two main directions:
1.technological research on fabrication of FA as a whole and of its constituent parts;
2.calculational research on substantiation of IRT-type FE and FA designs with regard to the WWR-SM reactor (Uzbekistan).
At present, large set of technological investigations of FE and FA fabrication processes and large set of neutronics, thermal hydraulic and strength calculations has been performed. These research of FE and FA design are being performed in close cooperation with NZChK, ANL and RRC KI within Russian and international RERTR programs.
The dispersion fuel U-9%Mo in Al matrix is chosen. The optimum U-235/FA loading is defined. Two full-scaled dummy of FA for hydraulic tests have been manufactured jointly by Bochvar Institute and Novosibirsk plant (NZChK).
Preliminary results of joint studies are presented in paper.

1. Introduction

IRT-type fuel assemblies with tube-type fuel elements are being used currently in six Russian-designed research reactors located abroad. At present there are several modifications of the IRT-type fuel assembly (IRT-2M, IRT-3M), which differ by the number and the size of fuel elements. To convert these rectors to LEU (19,75%), the IRT-4M tube-type fuel assembly has been developed on the basis of standard UO_2-Al fuel with uranium concentration - 3 g/cm^3 (~ 34,5 об.%). U-235 loading is equaled to 300 g in eight-tube FA, 265 g in six-tube FA [9]. Nevertheless, such loading does not permit utilization of this FA for conversion of some research reactors having power 10 MW (Uzbekistan, Libya). Therefore one of the main directions of Russian RERTR program is development of new fuel based on high-density U-Mo alloys. "TVEL" concern is coordinator of works of this program.

During the past few years this direction has been under intensive development. The use of new fuel can be accomplished in two designs of fuel assembly, namely in the currently used tube-type, and in a new pin-type.

The IRT-type FA with pin-type fuel elements is presented in this paper. The feasibility study of usage of FA based on dispersion U-Mo fuel is performed for the WWR-SM reactor located in the Uzbekistan where the irradiation tests of two full-scaled fuel assemblies are planned. The main object of this work is calculation-design development of FA, which will provide existing operating characteristics of a core.

The development are being performed in collaboration with NZChK, RRC KI and ANL within Russian and international RERTR program.

2. Fuel assembly design

The WWR-SM reactor core is currently used 18 fuel assemblies, placed on a square grid with step 71,5 mm (Fig.1). The core surrounded by a beryllium reflector has 9 horizontal beam tubes and more than 30 vertical irradiation channels. The core height is 600 mm. The reactor has been operated at a nominal power of 10 MW [8].

The IRT-3M type 6-tube FA with dispersion UO_2-Al fuel of 36% enrichment is used in the reactor. U-235 loading in FA is 309 g. This FA has the central channel either for control rod or for sample irradiation.

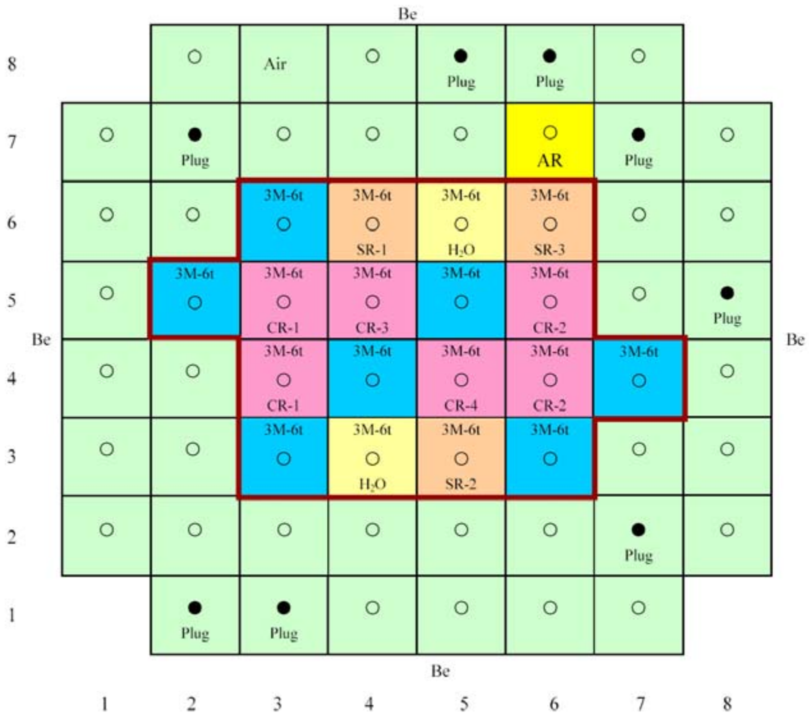
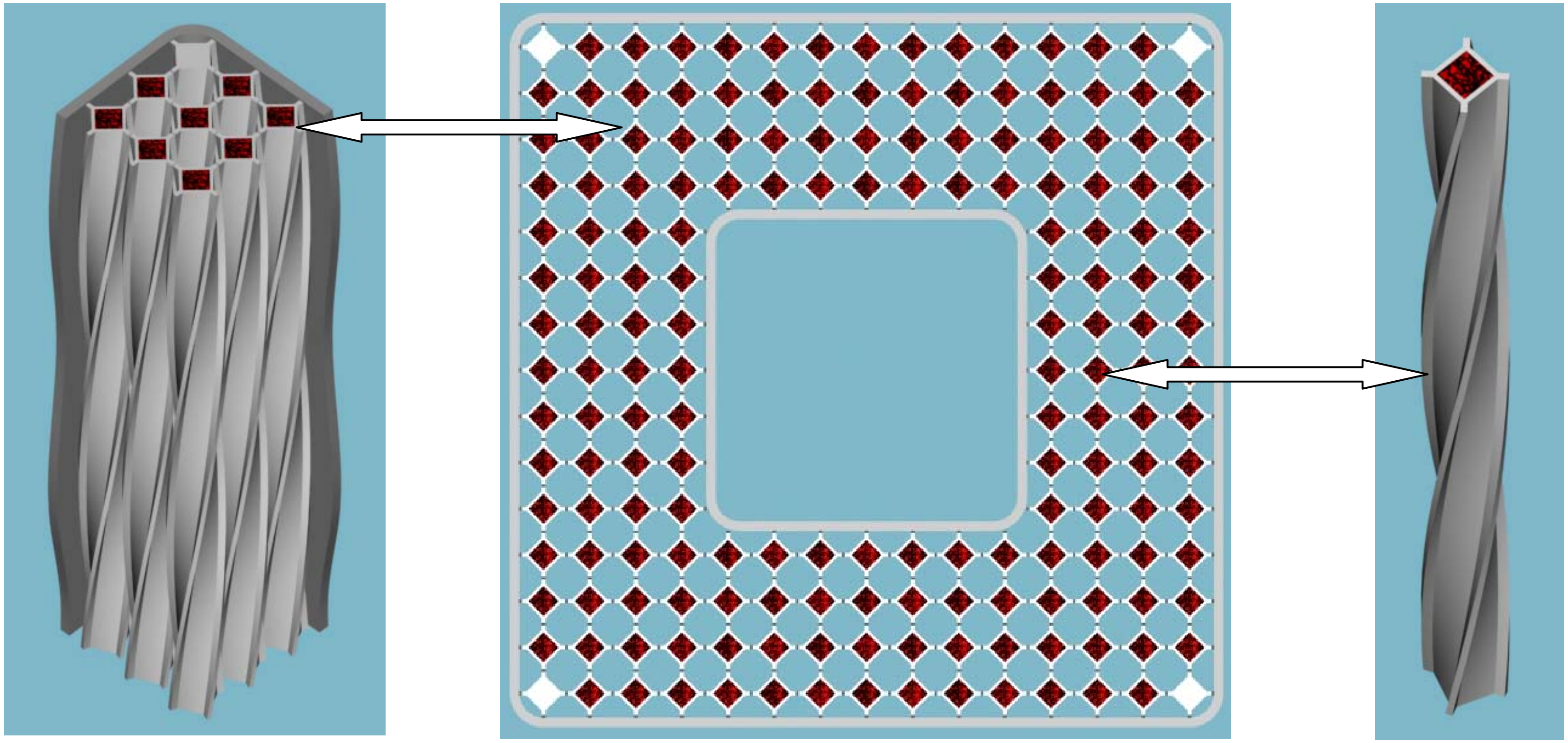


Fig.1. The WWR-SM core configuration

The principle design of IRT-type FA (Fig.2b) [2, 3, 6] with pin-type fuel elements consists of the end details and a square shroud, in which 176 pin-type fuel elements are placed by means of two spacer grids. The grids ensure the necessary attitude and arrangement step of fuel elements in FA. The ranging of the fuel elements is ensured by means of spiral ribs. The shape and the overall dimensions of shroud and channel for control rod are selected in the correspondence with the sizes of the elementary cell of the reactor cores and control rod. The sizes and design of the end details are preserved that's why it is possible to place pin-type fuel assembly in the core together with current type of fuel assemblies.

Fuel element is a square cross-section pin with four ribs curled around longitudinal axis (Fig.2c, [2]). The fuel element has the cladding, the dispersion meat and the end plugs. There is the diffusion contact between cladding, fuel meat and plugs. The aluminum alloys are used as constructional materials, and dispersion U-9%Mo fuel is used as nuclear fuel.



c) Arrangement of Fuel Elements in FA   a) Cross-section of FA   b) Pin-type fuel element

Fig.2. IRT-type FA with channel for control rod

The geometrical characteristics of the base design of fuel element and fuel assembly are given in Table 1. These characteristics were used in the technological investigations.

Table 1. The Geometrical Characteristics

Fuel Assembly	
Parameter	Value
Number of Fuel Elements in EFA	176
Area of Water Passage, mm^2	2079
Heat Transfer Surface, m^2	1,46
Specific Transfer Surface, 1/cm	4,9
Hydraulic Diameter, mm	2,89
Fuel Meat Volume, cm^3	465
Volume Fraction of Coolant	0,5

Fuel Element	
Parameter	Value
Step of Rib Spin, mm	320
Fuel Element Height/Meat, mm	645/600
Circumscribing Diameter, mm	4,5
Cladding Thickness, mm	min 0,3
Fuel Element Area, mm^2	8,89
Fuel Element Perimeter, mm^2	14,13
Fuel Composition Area, mm^2	4,41

3. Technological investigations

For during the past few years VNIINM accumulated a large technological and experimental experience of manufacturing of pin-type fuel elements. The received results of these developments allow to transfer the fabrication procedures to the plant for its finishing in industrial conditions. At present, in cooperation with the staff and under conditions of industrial production of the Novosibirsk plant (NZChK) a large set of technological investigations of FE and FA fabrication processes has been completed, namely:

- development of drawing-design and technological documentations;
- development of the design of spacer as a top and bottom grid;
- manufacturing of needed tools and rigs;
- development of fabrication process with fuel element simulators;
- development of fabrication process for outer and inner shrouds;
- development of the process of FA assembling.

The main share of works has been performed by the Novosibirsk plant.
As a result of aforementioned activities:

- two full-scaled dummy of FA for hydraulic tests have been manufactured jointly by Novosibirsk plant and Bochvar Institute;
- ability and preparedness for manufacturing of FA with pin-type fuel elements have been confirmed under conditions of currently operating industrial production.

4.Computational research

Computational research on substantiation of FE and FA design are being performed simultaneously with technological tests in close collaboration with ANL and RRC KI. The main goal of these calculations is to determine optimal parameters of pin-type FA, which will provide the best operating characteristics of the reactor. The core conversion variant is being selected on the base of following main criteria: 1) preservation of an equilibrium cycle length with the sufficient excess reactivity at the end of the cycle and the sufficient efficiency of CPS rods; 2) ensuring of rated power of the reactor.

Calculations have been carried out in two stages.

At the first stage the neutronics, thermal hydraulic and strength calculations has been performed for LEU core consisting of 18 fuel assemblies with pin-type base design of fuel elements (Table 1).

As a result of strength calculations the estimation of a limit loading at end part of FA design is made when it is charged/discharged. And also the FA-to-FA spacing in the core and the pin-to-pin spacing in FA have been justified.

The analysis of obtained data has shown, that:

- The rigidity (the absence of deflection failure) of the FA shroud is ensured when the axial loading is less or equal to $[P] \leq 400$ kg. The calculational model and results of calculations are given in Fig.3a. The distributions of stress and strain along height of shroud are shown in Fig. 3 (b, c) for axial load, which causes the deflection of FA shroud equaled to 1 mm.
- The load at the end parts of FA design is limited by collapse strength of boltless fastening. The allowable axial load should not exceed value equaled to $P \leq [P] = 280$ kg. At the same time the absence of deflection failure of FA shroud is ensured.
- Thus it is necessary to establish the limitation for axial load equaled to 280 kg. This value guarantees the strength of fastening of end parts of FA and the rigidity of the FA shroud when FA is inserted into or removed from core.
- The justification of the pin-to-pin spacing in FA was carried out on the basis of estimated calculation the mode of deformation and the change of fuel element shape. The calculations were carried out for cross- section of FE with the maximum burn-up of U-235. It is assumed that maximum burn-up is equaled to ~ 75%. It is obtained that the maximum change of diameter of FE on ribs is equaled to $\Delta D/D = 0{,}8\%$ (or the absolute value - 0,036 mm). Thus, to ensure the absence of standoff in a row of fuel elements and their pressure at FA shroud a technological pin-to-pin spacing in FA should not be less than 0,04 mm.

A large set of the neutron-physical calculations has been carried out by ANL and RRC KI. As a result the following input data were defined for thermal hydraulic calculations: U-235/FA loading, relative power of fuel assemblies, axial power density for each FE in FA.

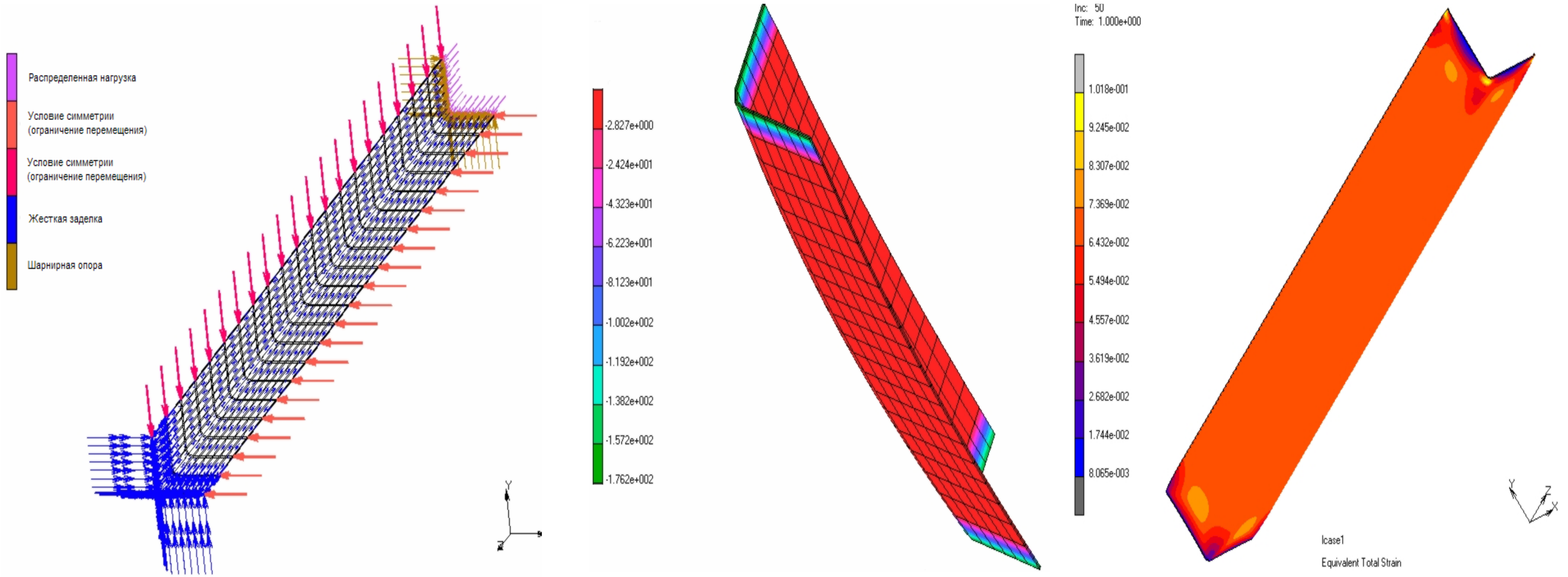
To estimate the peak power of reactor the multiple thermal hydraulic calculations of the most heated FA, located in the cell 6-5 (Fig.1) were performed. It was considered the most heated period of reactor operation - at the beginning of an equilibrium fuel cycle for the ^{135}Xe-Free core and typical positions of control rods in core of WWR-SM reactor at startup. Its duration is equaled to ~ 1-1,5 day. The calculations were performed with taking into account of power density of each FE in FA and inter-cell transfer in the FA cross-section (convection, turbulent and conductive). The two-dimensional heat transfer in FE cross-section is calculated using the numerical method of heat balances.

Computational model of FA cross-section is given in Fig.4.

The main limiting factor of fuel elements operation in this type of the reactor is the absence of surface boiling. The temperature for onset of nucleate boiling based on the Forster-Greif correlation [5] with an ONB margin of 1.4 is used as a limit factor.

To define the coolant flow through FA experimental correlation is used. This correlation was obtained in results the experimental studies of FA dummy consisting of 25 pins [4]. The fraction of the generated power in the fuel assemblies of the core is assumed to be 0.94.

The main results of the calculations for base design of FA are given in Fig. 5 and 6.



a) Calculational model

b) The distribution of stress along height of shroud (kg/mm^2) when axial load causes the deflection equaled to 1mm

c) The distribution of strain along height of shroud (kg/mm^2) when axial load causes the deflection equaled to 1mm

Fig.3. Calculation results of rigidity of FA shroud

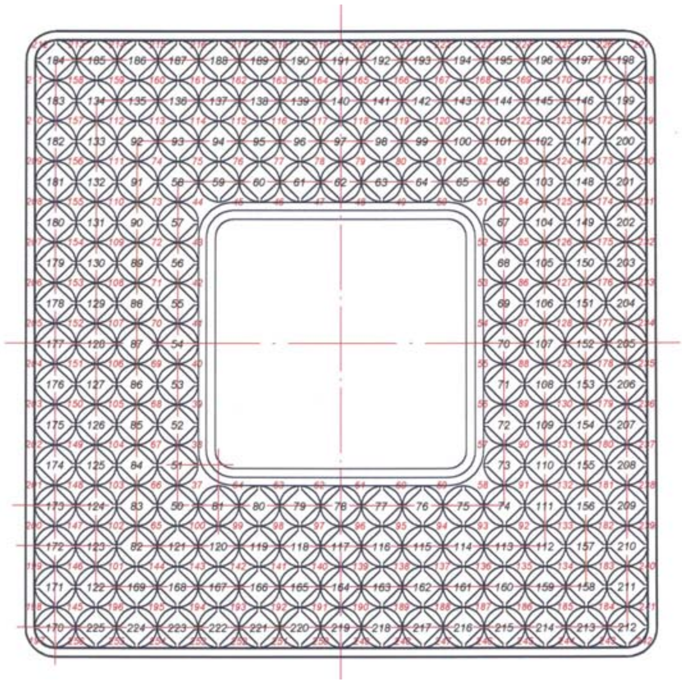


Fig.4. Computational model of FA cross-section

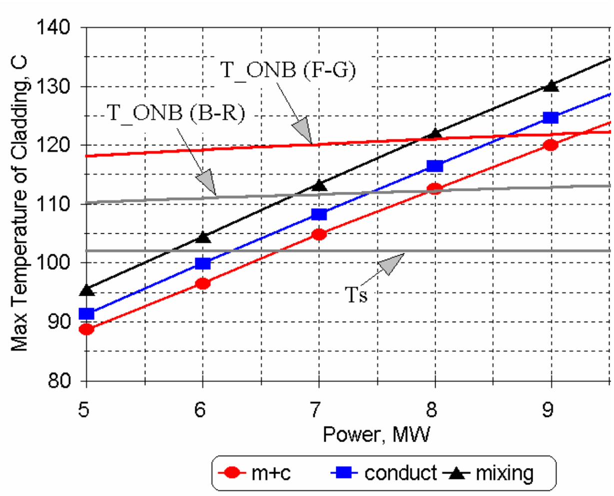


Fig.5. Change of cladding temperature of most heated FE N 198 vs reactor power for different models of inter-cell transfer.

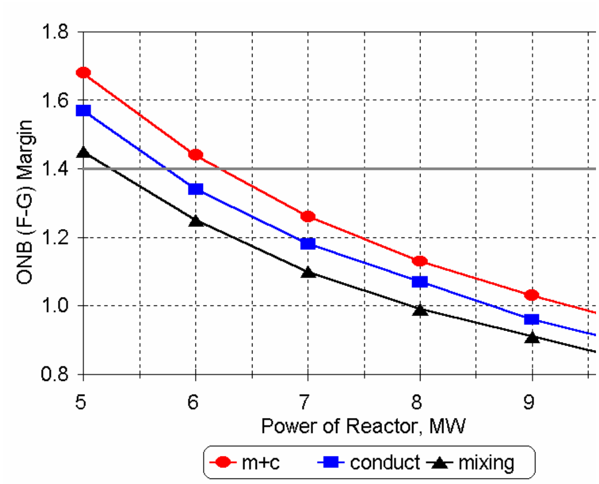


Fig.6. Change of ONB margin for most heated FE N 198 vs reactor power for different models of inter-cell transfer.

The analysis of the obtained results for base design of FA has shown, that:

- allowable power of reactor is defined by the cladding temperature of fuel elements located in corner of FA;
- at the beginning of an equilibrium fuel cycle the rated power of reactor should be decreased in ~ 30%.

To keep rated power of reactor at startup the second stage of calculations has been carried out. It is considered the following possible changes of design:

1. installing Al displacers in the corner pin positions;
2. change of FE size;
3. core configuration from 20 or 24 FA.

The neutronics and thermal hydraulic calculations have made for FA consisting of 172 fuel elements and 4 aluminum displacers in its corners. The size of FE meat is varied from 2.1 (base) up to 1.0 mm. It is considered three configurations of core from: 18, 20 and 24 FA.

The results of calculations for most heated FE in FA located in cell 6-5 are presented in Fig.17, 8, 9. In Fig. 7(a, b, c) are given the change of the minimum ONB margin for the most heated FE in FA verse power of reactor for core from 18, 20 and 24 FA accordingly. The change of the ONB margin along height of the most heated FE in FA at rated power of a reactor (10 MW) is shown in Fig.8 (a, b) for two configurations of core. The change of ONB (F-G) margin depending upon size of FE meat for two configurations of core: 18 FA and 20 FA is given in Fig.9 at rated power of a reactor (10MW).

The analysis of the obtained results for base design of FA has shown, that :

- Installation of Al displacers in corners of FA increases an allowable power of reactor in ~1,2 times.
- The using of the fuel assemblies of base design with size of FE meat of 2,1 mm with the installing of Al displacers in the corners of FA ensures the rated power of reactor (10MW) for core with 24 FA. The ONB margin is equaled to ~ 1,6.
- The reducing of the size of FE meat by increasing of rib height improves the hydraulic characteristics of FA that as a result an allowable power of reactor is appreciably increased.
- The using of the FE with meat size of ~1,5 mm in core from 18 FA and the FE with meat size of ~1,8 mm in core from 20 FA ensures the rated power of reactor (10MW).
- Using of FE design with monolithic UMo meat with size from 1,0 mm up to 1,3 mm ensures the rated power of reactor (10MW) for core from 18 FA with the ONB margin of ~ 1,45-1,5.

The final selection of the design of FE and FA will be made after completion of hydraulic tests of full-scaled FA dummy and thermal hydraulic calculations on its basis.

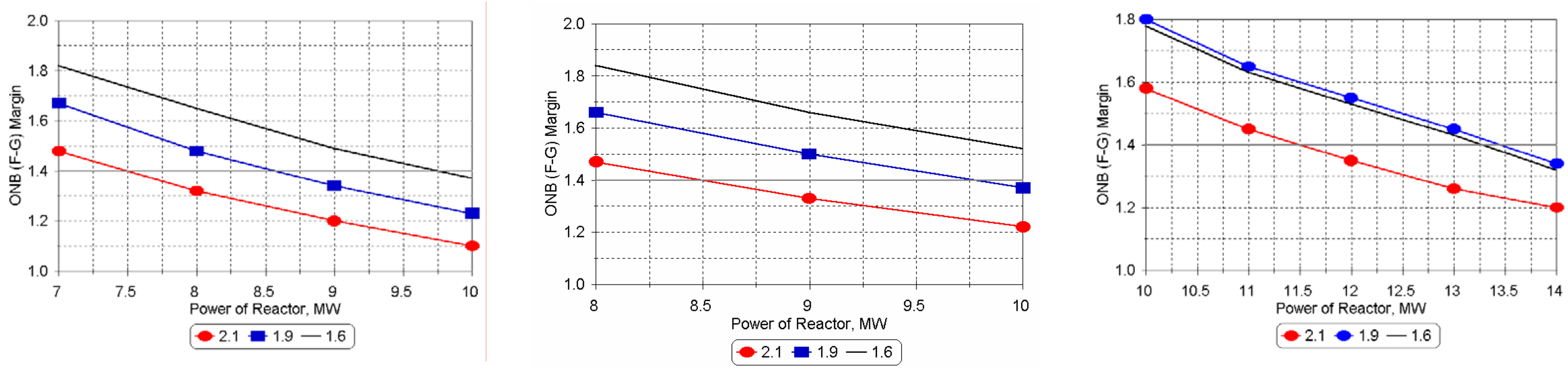
The hydraulic tests will be carried out on October 2003.

5. Conclusions

The design of an IRT-type FA with pin-type fuel elements based on LEU dispersion U-9%Mo fuel has been developed.

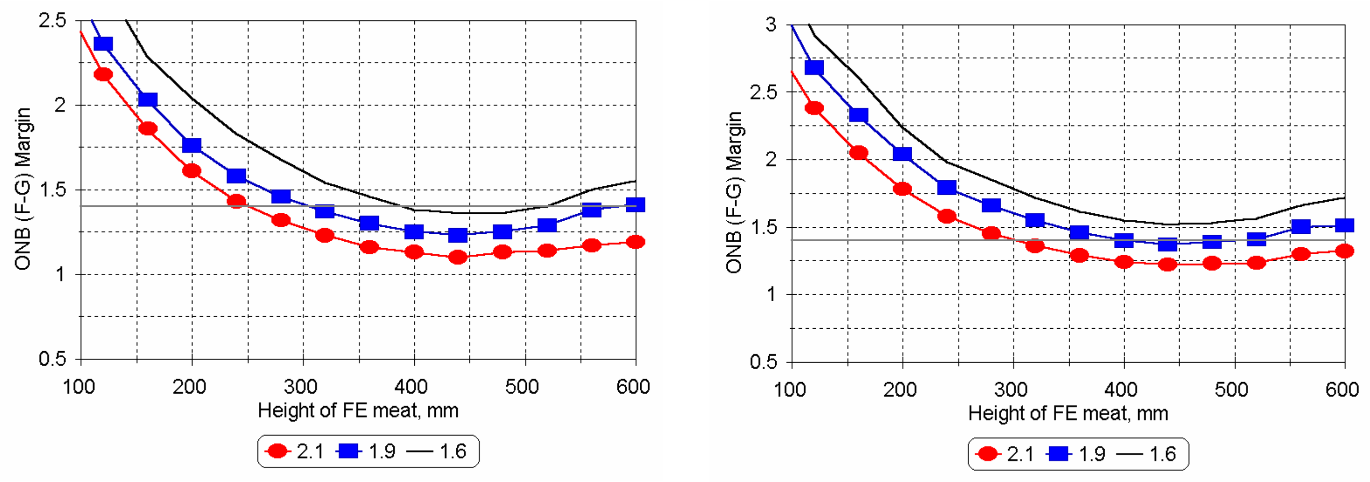
The development of FE and FA fabrication procedures has been completed under laboratory conditions and conditions of industrial production. Two full-scaled dummy of FA have been manufactured for hydraulic tests.

A large set of neutronics, thermal hydraulic and strength calculations for the WWR-SM reactor in Uzbekistan has been performed on substantiation of FE and FA design. The assessment of the data obtained will contribute to specification of the optimal parameters of dispersion pin-type fuel elements, also, the possibility of U-Mo monolithic fuel utilization has been shown.



a). 18 FA b). 20 FA c). 24 FA

Fig.7. The change of ONB margin (F-G) in most heated cross-section of the most heated FE



a). 18 FA b). 20 FA

Fig.8. The change of ONB margin (F-G) along height of the most heated FE (Nr=10 MW)

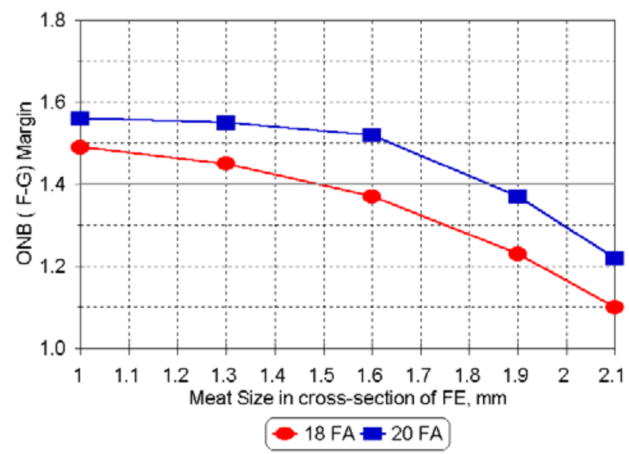


Fig.9. The change of ONB margin (F-G) vs size of FE meat (Nr=10 MW)

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References

[1] A.Vatulin, Y.Stetsky, I.Dobrikova. Unification of Fuel Elements for Research Reactors. 20th Int.Mtg. RERTR'97, Jackson Hole, Wyoming, USA, October 1997.

[2] A.Vatulin, Y.Stetsky, I.Dobrikova. Unified Fuel Elements Development for Research Reactors. 21st Int. Mtg. RERTR'98, Sao-Paulo, Brazil, October 1998.

[3] A.Vatulin, Y.Stetsky, I.Dobrikova, N. Arkhangelsky. Comparison of the parameters of the IR-8 reactor with different fuel assembly designs with LEU fuel. 3st Int. Mtg. RRFM'99, Bruges, Belgium, March 1999.

[4] A.Vatulin, Y.Stetsky, I.Dobrikova. A Feasibility Study of Using of the New Fuel Assembly Design For LEU Conversion of the IR-8 Research Reactor. 22nd Int. Mtg. RERTR'99, Budapest, Hungary, October 1999.

[5] Rapport CEA-R-4114, 1971.

[6] Report, describing the previous work on development of the prospective fuel element designed for the use of HDF (Delivery № 3.1).

[7] M.I.Solonin, A.V.Vatulin, Y.A.Stetsky, I.V.Dobrikova (VNIINM) and A.V.Ivanov, A.A.Kruglov, E.G.Leonov (Machine-Building Plant, Electrostal). Development of the new generation of fuel elements and fuel assemblies for research reactors. 12ve Int. Mtg. "Research Reactors: Science and High Technologies", Dimitrovgrad, Russia, 2001.

[8] A. Rakhamanov and etc. A Neutronic Feasibility Study For LEU Conversion of the WWR-SM Research Reactor in Uzbekistan. 21st Int. Mtg. RERTR'98, Sao-Paulo, Brazil, October 1998.

[9] Russian RERTR Program Works Status. 24th Int. Mtg. RERTR'2002, Bariloche, Argentina, November 2002.